10CFR50.73

Virginia Electric and Power Company
North Anna Power Station
P. O. Box 402
Mineral, Virginia 23117

April 25, 2000

U. S. Nuclear Regulatory Commission
Attention: Document Control Desk
Washington, D. C. 20555

Serial No.:	00-201
NAPS:	MPW
Docket No.:	50-339
License No.:	NPF-7

Dear Sirs:

Pursuant to 10CFR50.73, Virginia Electric and Power Company hereby submits the following Licensee Event Report applicable to North Anna Unit 2.

Report No. 50-339/2000-001-00

This report has been reviewed by the Station Nuclear Safety and Operating Committee and will be forwarded to the Management Safety Review Committee for its review.

Very truly yours,

W R Matthews

W. R. Matthews
Site Vice President

Commitments contained in this letter: None

Enclosure

cc: U. S. Nuclear Regulatory Commission
Region II
Atlanta Federal Center
61 Forsyth Street, SW, Suite 23T85
Atlanta, Georgia 30303

Mr. M. J. Morgan
NRC Senior Resident Inspector
North Anna Power Station

JE22

NRC FORM 366 (6-1998)

U.S. NUCLEAR REGULATORY COMMISSION

APPROVED BY OMB NO. 3150-0104
EXPIRES 06/30/2001

LICENSEE EVENT REPORT (LER)

(See reverse for required number of digits/characters for each block)

FACILITY NAME (1)	DOCKET NUMBER (2)	PAGE (3)
NORTH ANNA POWER STATION , UNIT 2	05000339	1 OF 6

TITLE (

AUTOMATIC REACTOR TRIP DUE TO LOSS OF THE STATION SERVICE TRANSFORMER

EVENT DATE (5)			LER NUMBER (6)			REPORT DATE (7)			OTHER FACILITIES INVOLVED (8)	
MONTH	DAY	YEAR	YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	MONTH	DAY	YEAR	FACILITY NAME	DOCUMENT NUMBER 05000-
04	03	2000	2000	001	00	04	25	2000	FACILITY NAME	DOCUMENT NUMBER 05000-

		THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: (Check one or more) (11)							
OPERATING MODE (9)	1		20.2201(b)		20.2203(a)(2)(v)		50.73(a)(2)(i)		50.73(a)(2)(viii)
POWER			20.2203(a)(1)		20.2203(a)(3)(i)		50.73(a)(2)(ii)		50.73(a)(2)(x)
LEVEL (10)	100%		20.2203(a)(2)(i)		20.2203(a)(3)(ii)		50.73(a)(2)(iii)		73.71
			20.2203(a)(2)(ii)		20.2203(a)(4)	X	50.73(a)(2)(iv)		OTHER
			20.2203(a)(2)(iii)		50.36(c)(1)		50.73(a)(2)(v)		Specify in Abstract below
			20.2203(a)(2)(iv)		50.36(c)(2)		50.73(a)(2)(vii)		or in NRC Form 366A

LICENSEE CONTACT FOR THIS LER (12)

NAME	TELEPHONE NUMBER (Include Area Code)
W. R. Matthews, Site Vice President	(540) 894-2101

COMPLETE ONE LINE FOR EACH COMPONENT FAILURE DESCRIBED IN THIS REPORT (13)

CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO EPIX	CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO EPIX
X	EA	CBL5	C803	Y	X	AA	ZI	E001	N
X	SN	MO	S905	N	X	AA	LF	W120	N

SUPPLEMENTAL REPORT EXPECTED (14)				EXPECTED SUBMISSION DATE	MONTH	DAY	YEAR
	YES (If yes, complete EXPECTED SUBMISSION DATE).	X	NO				

ABSTRACT (Limit to 1400 spaces, i.e., approximately 15 single-spaced typewritten lines) **(16)**

On April 3, 2000, at 0857 hours, with Unit 2 in Mode 1 operating at 100% power, an automatic reactor trip occurred as a result of actuation of the 2C Station Service Lockout Relay resulting in a Turbine trip-Reactor trip. Actuation of the 2C Station Service Lockout Relay was due to a fault on a secondary feeder cable from the 2C Station Service Transformer to the 2C Station Service Bus. Control Room personnel responded to the reactor trip in accordance with emergency procedure E-0, Reactor Trip or Safety Injection. Initially, Reactor Coolant System (RCS) pressure decreased to 1985 psig, pressurizer level decreased to 23 percent, and RCS temperature decreased to approximately 542 degrees F. Pressurizer pressure, level and RCS temperature returned to their normal programmed values. All Engineered Safety Feature (ESF) equipment responded as designed. This event posed no significant safety implications because the reactor protection system and ESF systems functioned as designed following the reactor trip. Therefore, the health and safety of the public were not affected by this event.

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LICENSEE EVENT REPORT (LER)
FAILURE CONTINUATION

FACILITY NAME (1)	DOCKET NUMBER (2)	LER NUMBER (6)			PAGE (3)
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COMPLETE ONE LINE FOR EACH COMPONENT FAILURE DESCRIBED IN THIS REPORT (13)

CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO EPIX	CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO EPIX
X	SB	TCV	C635	N					

1.0 DESCRIPTION OF THE EVENT

On April 3, 2000, at 0857 hours, with Unit 2 in Mode 1 operating at 100% power, the 2C Station Service Lockout Relay (EIIS System EA, Component 86) actuated resulting in an automatic Turbine trip-Reactor trip. The relay actuation was due to a low side feeder cable (EIIS Component CBL5) failure (4160 volt lead) from the 2C Station Service Transformer (SST) (EIIS Component XFMR) to the 2C Station Service Bus (EIIS Component BUS).

Control Room personnel responded to the reactor trip in accordance with emergency procedure E-0, Reactor Trip or Safety Injection. Initially, Reactor Coolant System (RCS) (EIIS System AB) pressure decreased to 1985 psig, pressurizer level decreased to 23 percent, and RCS temperature decreased to approximately 542 degrees F. Pressurizer pressure, level and RCS temperature returned to their normal programmed values and Unit 2 was brought to a safe stable condition. All Engineered Safety Feature (ESF) equipment responded as designed. The post trip response progressed as expected and the Operators transitioned to 2-ES-0.1, Post Trip Recovery. The plant was stabilized at no-load conditions.

The Auxiliary Feedwater System (AFW) (EIIS System BA) received an auto start signal as expected for a full power reactor trip. The unit equipment responded as expected with a few discrepancies. A non-emergency four hour report was made to the NRC Operations Center at 1002 hours on April 3, 2000, pursuant to 10 CFR 50.72(b)(2)(ii) for an event that resulted in an automatic actuation of any engineered safety feature including the reactor protection system.

At 1014, the Unit 2 AFW system received another auto start, after initially being secured, due to the 'C' Steam Generator (S/G) (EIIS System AB, Component SG) level decreasing below the low-low level setpoint. At 1047 hours the AFW pumps (EIIS System BA, Component P) were secured and placed in automatic with S/G levels approximately 40 percent. A second non-emergency four hour report was made to the NRC Operations Center at 1052 hours on April 3, 2000, pursuant to 10 CFR 50.72(b)(2)(ii).

At 1028, the 'B' Steam Dump valve (EIIS System SB, Component TCV) failed open. The valve had been modulating properly immediately following the unit trip at 0857. When the valve failed, steam dump control was in steam pressure mode. The steam dump switches were placed in the off position, and the 'B' Steam Dump valve closed after a delay of approximately 20 seconds. The instrument air supply was then isolated to the 'B' Steam Dump and the steam dumps were placed back in the steam pressure mode. Even with the 'B' Steam Dump valve isolated, the steam dumps controlled temperature without any problems.

Also during the trip the 'A' High Pressure (HP) Heater Drain pump motor (EIIS System SN, Component MO) developed an oil leak. The discharge check valve was inspected for possible back leakage, which may have caused a reverse rotation of the motor. The inspection did not identify any discrepancies with the check valve.

1.0 DESCRIPTION OF THE EVENT (continued)

Following the Unit 2 reactor trip, Individual Rod Position Indicator's (IRPI) (EIIS System AA, Component ZI) for control rods, N9, H14, G13, and D4, were indicating greater than 10 steps withdrawn with K6 and L11 initially indicating approximately 170 steps with no rod bottom light (EIIS System AA, Component LF) illumination for K6.

The Control Room did not receive indication of the 2C Moisture Separator Reheater inlet flow control valve (EIIS System SB, Component FCV) going closed.

Withdrawal of the shutdown bank control rods started at 0403 on April 4, 2000. At 0557, Mode 2 was entered and Unit 2 went critical at 0624 hours on April 4, 2000.

2.0 SIGNIFICANT SAFETY CONSEQUENCES AND IMPLICATIONS

This event posed no significant safety implications because the reactor protection system and ESF systems functioned as designed following the reactor trip. Therefore, the health and safety of the public were not affected by this event.

These events are reportable pursuant to 10 CFR 50.73 (a)(2)(iv) for a condition that resulted in an automatic actuation of any engineered safety feature including the reactor protection system.

3.0 CAUSE

The cause of the automatic reactor trip is the due to actuation of the 2C Station Service Lockout Relay resulting in a Turbine trip-Reactor trip. Actuation of the 2C Station Service Lockout Relay was due to a fault on a 4160 volt low side cable that shorted to the supporting cable tray. Cause of the 4160 low side cable failure has not yet been determined.

The cause of the second automatic start of the AFW system was due to the 'C' Steam Generator (S/G) level decreasing below the low-low level setpoint. Following a reactor trip redundant components not required for hot standby (Mode 3) operation are secured. To shutdown the redundant condensate pump the discharge valve for the condensate pump (EIIS System SD, Component P) was being throttled closed. At the same time, feed control to the "C" S/G was complicated by the demand indication of the "C" main feedwater bypass feed regulating valve (EIIS System SD, Component FCV) which was lagging behind the "A" and "B" bypass feed regulating valves. An alarm for condensate pump discharge header low pressure was received and Operation's personnel tried to re-open the discharge valve. At this point S/G level decreased below the auto start setpoint of the AFW system.

4.0 IMMEDIATE CORRECTIVE ACTION(S)

Control Room personnel responded to the reactor trip in accordance with emergency procedure E-0, Reactor Trip or Safety Injection. Initially, RCS pressure decreased to 1985 psig, pressurizer level decreased to 23 percent, and RCS temperature decreased to approximately 542 degrees F. Pressurizer pressure, level, and RCS temperature returned to their normal programmed values and Unit 2 was brought to a safe stable condition. All ESF equipment responded as designed.

The post trip response progressed as expected and the Operators transitioned to 2-ES-0.1, Post Trip Recovery. The plant was stabilized at no-load conditions.

5.0 ADDITIONAL CORRECTIVE ACTIONS

A Post Trip Review meeting was conducted at 1100 hours on April 3, 2000, with station personnel to identify the cause of the reactor trip to prevent recurrence, to identify abnormal or degraded indications occurring during the reactor trip, and to assess Unit readiness for return to operation.

The failed 4160 volt low side cable was repaired and the 2C SST was returned to service on April 5, 2000.

The "A" HP heater drain pump motor was replaced and the pump was returned to service on April 6, 2000.

The IRPIs for control rods, D4, H14, G13, and N9 were indicating greater than 10 steps withdrawn. In addition, the IRPIs for K6 and L11 initially indicated approximately 170 steps with the rod bottom light not lit for K6 and the SPDS and PCS indicating K6 and L11 were less than 10 steps withdrawn. The as found calibration data showed that all of these indicators were within 6 steps of desired (zero or rods full in). The rod bottom bi-stable set point for control rod K6 was checked and found at 19.7 steps with a desired of 20 +/-2 steps. The calibration data shows that all of these IRPIs were well within the desired limits. The cause of the 170 step indication for IRPI K6 and L11 was a stuck needle. The indicator scale had shifted and was pressing against the needle causing it to stick at about 170 steps. Failure of the control rod K-6 bottom light to illuminate was the result of a loose light socket preventing it from coming on when the control rod reached the bottom. Repairs were made and the rod bottom light functions properly. Rod drop testing was also performed satisfactorily for control rods K6 and L11.

The "B" steam dump valve maintenance was completed and the valve returned to service on April 11, 2000.

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TEXT (If more space is required, use additional copies of NRC Form 366A) (17)

The Control Room indication for 2C Moisture Separator Reheater inlet flow control valve was determined to be an incorrect limit switch (EIIS System SB, Component ZIS) position at the valve. The limit switch was adjusted and Control Room indication was established.

6.0 ACTIONS TO PREVENT RECURRENCE

A root cause evaluation is being performed regarding the automatic reactor trip including the second AFW actuation. Corrective actions will be performed as necessary following completion of the evaluation.

7.0 SIMILAR EVENTS

No events were identified for an automatic reactor trip as a result of a SST failure.

8.0 ADDITIONAL INFORMATION

Unit 1 was in Mode 5 for a scheduled refueling outage.

Component failure information:
Description: Electrical cable
Manufacturuer: Anaconda
Model No.: NGA-08

Description: HP Heater Drain Pump Motor
Manufacturuer: Siemens Energy
Model No.: GV

Description: Individual Rod Position Indicator
Manufacturuer: EIL Instruments
Model No.: EIL-P-1153A

Description: Rod Bottom Light Socket
Manufacturuer: Westinghouse

Description: Steam Dump Valve
Manufacturuer: Copes-Vulcan
Model No.: D-100-160-3